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SIMULATION OF THE BURNUP IN CELL CALCULATION USING THE WIMSD-5B CODE CONSIDERING DIFFERENT NUCLEAR DATA LIBRARIES

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ABSTRACT

This work proposes to implement the cell calculation considering the fuel burning using the WIMSD-5B code. The cell calculation procedure allows to determine the nuclear parameters present in the multi-group neutron diffusion equation and for this purpose the neutron transport theory is used in a problem with dimensional reduction, but in contrast is considered a large number of groups associated with the neutron spectrum. There are a variety of reactor physics codes that determine the nuclear parameters by solving the neutron transport equation applied to an equivalent cell representing a fuel element. The WIMSD-5B code is a deterministic code that solves the transport equation using collision probability method. The simulation of fuel burning in the cell calculation took into account different nuclear data libraries. The WIMSD-5B code supports several nuclear data libraries and in the present work the following libraries were used: IAEA, ENDFB-VII.1, JENDL3.2, JEFF3.1 and JEF2.2, all formatted for 69 energy groups.

1. INTRODUCTION

In a first analysis of a reactor core there are computational implementations that consider the nucleus with infinite dimension, exploring repetitive geometry, and a high number of energy groups. These computational implementations are called cell or network calculation codes. These codes are used to determine the distribution of the neutron flux and the multiplication factor in an infinite medium. For this it is necessary to provide with input data from these codes the multi-group isotope nuclear data sets and also the description of the cell that repeats infinitely in the reactor, also known with equivalent cell.

The cell calculation codes solve the neutron transport equation over a specific region of the reactor. This region can be a unit cell or a macro-cell [1]. Therefore, cell calculation codes include methods to solve an appropriate set of equations for neutron flux and infinite multiplication factor, k_∞, considering discretized energy and a spatial network (energy groups and discretized spatial points). The calculated neutron flux can be used to obtain sets of homogenized macroscopic cross sections covering selected sub regions and in a broad energy group structure. These sets of macroscopic cross sections are then used as input material data for several codes that solve the neutron transport equation or the neutron diffusion equation over the entire reactor core or part thereof. The neutron flux thus calculated can then be used

for the calculation of reaction rates or in fuel depletion calculations distributed over this effective nucleus.

Another important feature that is found in most cell calculation codes and the ability to simulate firing during the operation of the nuclear reactor, providing the concentrations of the various nuclides as well as the sets of macroscopic cross sections at each firing step .

2. WIMSD-5B CODE

The Winfrith Improve Multigroup Scheme (WIMS) is a deterministic code based on the transport theory to calculate the flux as a function of the energy and position in the cell, performing calculation for different geometries, providing the physical parameters necessary for the development of reactor designs nuclear weapons of various types. Since the 60's it has been successful in its results, in addition it is widely accepted internationally, being one of the most used in the management of core nuclear reactors. Its standard library has data from multi-group cross sections of the materials most commonly used in nuclear reactors.

Among the various versions of WIMS can be highlighted: LWRWIMS, WIMSD and WIMSE. WIMSD versions are not commercial in nature and are available from the Nuclear Energy Agency (NEA) Data Bank. The code in all versions was developed for a library with a maximum number of 69 energy groups, whose structure allows an extension of the number of nuclides and datasets of any isotope.

The code WIMSD-5B uses the nuclear data libraries with the same structure of 69 energy groups of the original code, WIMS-D, with 14 fast groups between 9,118 keV and 10 MeV, 13 groups of resonances between 4 eV and 9,118 keV, and 42 thermal groups from 0 to 4 eV.

More recently, with new updates, the code now has libraries with up to 172 power groups or more. Among the various available libraries may be cited the following: WDN29, ENDFB-VII.1, ENDFB-VII.1GX, JENDL3.2, JENDL3.2GX, JEF2.2, JEF2.2GX, JEFF3.1, JEFF3.1GX, IAEA and IAEAGX [2]. The WIMSD code develops the calculation of cells in four different geometries: homogeneous cells, cylindrical plates or bars, bar or plate arrays (cluster geometry) and multi-cells. In this work the calculation was made from the data of a one-dimensional cell (Slab).

Some of the most significant items produced by the calculations of this code are: the infinite multiplication factor, k_∞, the effective multiplication factor, if the leakage data, isotope reaction rates present in the calculation are reported, the mean constants and the final concentrations of the nuclides and fission products in cases of burning [3].

2.1. Burning Calculation in WIMSD5B code

In the WIMSD code the burning equation for each material and isotope *i* is written as follows:

$$\frac{dN_i(t)}{dt} = -(\lambda_i + \{RR\}_i^a)N_i(t) + \sum_k (t)\, q_{i,k'}(t)N_k(t) + \sum_k q_{i,k}(t)\, N_k(t) \qquad (1)$$

where:

λ_i = decay constant of the nuclide i;
$N_i(t)$ = concentration of the nuclide i;
$\{RR\}_i^a$ = the rate of absorption reaction of the nuclide i;
q_{ik} = are the production terms calculated from the yield of the fission products i by the fission of the nuclide k, production fractions, gamma capture and fission reaction rates.

2.2. General Sequence of WIMSD Code Calculation

The description of the code can be summarized in four steps: data preparation, central calculation, editing and calculation of firing. First, using the SPECTROX collision theory method [4], the code calculates the spectrum for some spatial regions using its entire energy group library (69 or 172 groups) and uses that spectrum to collapse the cross sections for the number of groups that will be taken to the next step. In the central calculation, the transport equation is solved for a given energy group and for a specific geometry using the DSN or collision probability methods. At the editing stage, corrections are made to the data obtained in the previous step, such as leak correction, effective multiplication factor, buckling that makes the system critical, and rate of reactions. Finally the burn calculation is performed. After completing these steps, the process can be repeated for the initial phase or shut down.

3. EVALUATED NUCLEAR DATA LIBRARIES

There are numerous evaluated nuclear data libraries available from various nuclear data centers. National interests and different applications are the two main factors that cause this variety. Countries with strong nuclear programs, such as the US, the European Union, Japan, Russia and China, develop their own general purpose libraries to maintain assessment expertise and ensure technological independence. On the other hand, various applications of nuclear technology require specific purpose libraries that satisfy the particular needs of a given application [5].

These derived libraries add another class to the one mentioned above. It should also take into account several versions (releases) of the main libraries. The frequent sharing of evaluations between different libraries, often with some modifications, makes this picture even more complicated [5].

General-purpose libraries are not limited to any specific application and are intended to satisfy a wide class of users. In practice, however, they have often emerged as libraries for specific reactor applications. Evaluations in a general purpose library are generally more complete in terms of physical quantities and nuclear reactions. They must be suitable for transport calculations and as such must meet very stringent requirements for integrity and consistency [5].

Thus, evaluations of neutron behavior have to cover the resolved, unresolved thermal resonance as well as the gamma-ray of fast neutrons that extend at least up to 20 MeV, and should contain all major reaction channels, provide cross and possibly angular distributions, cross sections correlated with energy angle, and photon production data. The internal consistency implies that the individual cross sections must add the total cross sections and the

integral emission spectra corresponding to the respective cross section of the reaction. Typically, general purpose libraries are extensively validated against integral measurements [5].

Sometimes the results of these integral measures are incorporated into a library. This procedure introduces implicit correlations between various reactions and materials causing that library to become an entity rather than a simple collection of individual evaluations. The main general purpose libraries are maintained by the following countries:

1. USA - ENDF / B-VII.1, launched in 2011;
2. Europe - JEFF-3.1, launched in 2005, JEF-2.2, launched in 1992;
3. Japan - JENDL-3.3, launched in 2002, JENDL-4.0, launched in 2010;
4. Russia - BROND-2.2, launched in 1992; BROND-3 has not yet been completed, is partially available in selected ROSFOND assessments, which was launched in 2008;
5. China - CENDL-2, launched in 1991; CENDL-3 was developed, but not internationally released; Development of CENDL-4 is in progress.

During the first decade of the 21st century, three nuclear data libraries evaluated (ENDF / B, JEFF and JENDL) were continually updated and improved. These libraries will be briefly summarized later in this section.

3.1. The ENDF-6 Format

The use of the ENDF-6 format is common for most of the nuclear data libraries evaluated. Only some of the activation and derived libraries deviate from this pattern. Otherwise, all libraries are using the internationally accepted ENDF-6 format. This unification had a major impact on global cooperation, facilitated by the exchange of files between national libraries and the easy comparison of data. The ENDF format was developed by the CSEWG and is maintained by the NNDC.

The work began in 1966, the first version was released in 1968, then in 1970, 1972, 1975, 1979 and 1990 along with the subsequent releases of the US ENDF library. The current version, ENDF-6 [6], has been used for both the ENDF / B-VI and ENDF / B-VII libraries, implying that a new version of the format was not developed for the ENDF / B-VII. We note that to distinguish it from the library that is indicated with Roman numerals (say, ENDF / B-VI), the ENDF format is always denoted by the Arabic numeral (ENDF-6).

For historical reasons, the ENDF-6 format uses 80-character records according to the old versions of FORTRAN. It is organized in a strict hierarchical structure. Any library is a collection of material evaluations from a recognized assessment group. The ENDF-6 format is divided into sub-libraries that distinguish between different incident particles and data types, namely, neutron-induced reactions, proton-induced reactions, thermal dispersion data, fission yields, decay data, etc.

An evaluation is the process of analyzing experimentally measured physical parameters (such as cross sections), combining them with predictions of nuclear model calculations, and attempting to extract the true values of such parameters. The process of parameterizing and reducing data to the tabular form produces a data set evaluated.

The sub-libraries contain the data for different materials identified by MAT numbers. Each material evaluation contains blocks of data called "Files" and indicated with MF numbers.

3.2. The ENDF/B Library

The ENDF/B-VII.0 library, released by the US CSEWG in December 2006, contains data primarily for reactions with incident neutrons, protons, and photons on almost 400 isotopes, based on experimental data and theory predictions. The new library plays an important role in nuclear technology applications, including transport simulations supporting national security, nonproliferation, advanced reactor and fuel cycle concepts, criticality safety, fusion, medicine, space applications, nuclear astrophysics, and nuclear physics facility design.

The Cross Section Evaluation Working Group (CSEWG) launched the ENDF/B-VII.1 library on December 22, 2011. The ENDF/B-VII.1 library is the latest core data file rated and recommended for use in applications of nuclear science and technology, and incorporates advances in the five years since the release of ENDF / B-VII.0, including: many new assessments in the neutron sub-library (423 in total) and more than 190 of these contain covariates, new yields of fission products and a large expansion of decay data [5].

3.3. The JEFF-3.1 Library

The JEFF project is a collaborative effort among the European member countries of the NEA Database. The initial objective was to improve the performance of existing reactors and fuel cycles. More recently, the goal is to provide users with a more comprehensive set of data for a wide range of applications, including the innovative reactor concept, radioactive waste transmutation, fusion and medical applications. These data include reactions induced by neutrons and protons, radioactive decay, fission yields, thermal scattering law and photo-atomic interactions.

The JEFF-3.1 version of the library was released in May 2005, for a brief description see JEFF Report. 19 [7]. The library combines the efforts of the JEFF and EFF / EAF (European Fusion File / European Activation File) work groups. The general purpose neutron sub library contains 381 materials from 1 to ^{255}Fm. The activation sub-library is based on EAF-2003 and contains cross sections for neutron reactions in 774 targets; The radioactive decay sub library contains three 852 isotopes of which only 226 are stable; The proton sub library covers 26 materials from ^{40} Ca to ^{209} B; The thermal scattering law sub-library includes nine materials; The neutron induced fission yield sub-library covers 19 isotopes of ^{232}Th at ^{245}Cm, and the spontaneous fission yield sub library of $^{242, 244}$Cm and ^{252}Cf.

The JEFF-3.0 library was improved mainly due to the sub prediction of the reactivity of LEU systems relevant to light water reactors. The question of reactivity was linked to the ^{238}U cross sections and the improved assessment was obtained as a result of extensive international effort. The transport calculations showed that the predictions of this reactivity were appreciably improved.

New evaluations or major revisions were performed for Ti isotopes (IRK Vienna); Ca, Sc, Fe, Ge, Pb and Bi isotopes (NRG Petten); Isotopes ^{103}Rh, $^{127, 129}$I and Hf; E $^{236, 237, 238}$ U and ^{214} Am (CEA). For other isotopes, more recent evaluations of other libraries have been adopted. Revised thermal spreading data were produced for all important moderator and structural materials.

The JEFF project has put considerable effort into validating the library, which has been done with special care from the point of view of the applications of nuclear reactors. The overall performance of the library is excellent.

3.4. JEF-2.2 Library

The library version of the general purpose library was launched in January 1992. The library contains neutron reaction data evaluations for 313 elements or isotopes from 1H to 99Es in neutron energy ranging from 10E-5 eV to 20 MeV, in the format ENDF-6. Although it was a library that was not very updated, it was formatted to be used in WIMSD-5B code [2].

3.5. JENDL-3.3 and JENDL-3.2 Library

JENDL-3.3 is the Japanese library evaluated that was launched in 2002, see the summary of [8]. The library relies heavily on assessments that originated in Japan, probably representing the most extensive source of independent evaluations, shortly after the US effort.

The goal of the JENDL effort is to provide Japanese data evaluated for fast breeding reactors, thermal reactors, fusion neurons, and shielding calculations as well as other applications. The JENDL-3.3 library contains data for 337 materials, from 10-5 to 20 MeV.

The main issues of the previous version of the library, JENDL-2.2, were addressed: the overestimation of the criticality values for the thermal fission reactors was improved by modifying the fission cross sections and fission neutron spectra for the ^{235}U; Incorrect energy distributions of minor neutrons of major heavy materials were replaced by the results of a statistical model; Inconsistency between elemental and isotopic evaluations were removed for the heavy nuclides.

The JENDL-4.0 version of the library was released in 2011. Only the JENDL-3.2 version was formatted for the WIMD-5B code [2].

3.6. IAEA Library

The idea of a project to upgrade the WIMS multi-core nuclear data entry library came from discussions among participants, speakers and directors of the IAEA / ICTP Joint Workshop on Nuclear Technology Reactor Physics Calculations conducted at ICTP, Trieste , from 12 February to 16 March 1990. This idea was in line with the relevant conclusions and recommendations of the meeting of the Technical Committee Meeting on In-Nuclear Fuel Management conducted by the IAEA in Vienna from 4 to 7 December 1989 [2].

The WIMS Library Update Project (WLUP) was organized by the IAEA Nuclear Data Section. The Jozef Stefan Institute was supported by the IAEA through a research contract to

coordinate technical issues related to the project. It should be noted that, except for coordination work and some IAEA fellowships, participants contributed to the project on a voluntary basis.

The Nuclear Data Technology Subcommittee of the Nineteenth INDC Meeting, March 8-12, 1993, strongly endorsed the continuation of the WIMS Library Update Project to provide updated multi-group cross sections in the WIMS-D format for thermal research and power reactors and stated that this project was also in line with the Agency's technology transfer objectives.

The project was also recommended by participants from the IAEA Consultants Meeting on Nuclear Data Processing conducted in 1993 and from developing country experts who participated in the IAEA / ICTP Workshop on Calculations, Design and Safety of Reactors in 1994.

In 1996, the IAEA Division of Physics took the initiative to create the WLUP as a Coordinated Research Project (CRP). This organizational scheme accelerated the completion work of the updated WIMS-D library and allowed its extensive validation for network calculations for thermal reactors.

The final phase of the WLUP as Coordinated Research Project (CRP) began in December 1998, with contracts or research agreements for participants from Argentina, Bangladesh, China, Cuba, India, Korea, Morocco, Poland, Russia, Slovenia, and the United States. . Research Coordination Meetings were held in Vienna, Austria (1999), Bariloche, Argentina (2000) and Vienna, Austria (2001).

Upon completion of WLUP, the following products are available for free to WIMS-D users:

1. Cross Section Library with 69 IAEA-WIMSD groups: includes over 170 selected nuclear data file materials evaluated in a 69-group energy structure. The library has been extensively validated in more than 200 reference cases;
2. Cross Section Library with 172 IAEA-WIMSD groups: Includes over 170 materials from selected nuclear data files selected in a power structure of 172 groups. The library has been extensively validated in more than 200 reference cases;
3. IAEA-TECDOC with detailed documentation;
4. NJOY for processing of assessed nuclear data;
5. "Benchmarks" of entries for the WIMS-D code;
6. Benchmarks of results for the libraries considered in WLUP;
7. Auxiliary codes and procedures for updating and maintaining the WIMS-D library;
8. WIMS-D updates to extend code capabilities.

The libraries mentioned in item 6 are as follows: ENDFB-VII.1, ENDFB-VII.1GX, ENDFB-VII.0, ENDFB-VII.0GX, ENDFB-VI.0, ENDFB-VI.0GX, JENDL3.2, JENDL3.2GX, JEF2.2, JEF2. 2GX, JEFF3.1 and JEFF3.1GX. "GX" means that they are nuclear data libraries evaluated considering 172 power groups.

4. RESULTS

In order to simulate burning using the WIMSD-5B code, it was considered the MTR-IAEA research reactor of the benchmark problem conceived by the IAEA. This research reactor consists of plate-type fuel elements. In the 1980s the IAEA coordinated a work with the collaboration of different organizations to help reactor operators determine the feasibility of converting high uranium enrichment (HEU) reactors to low uranium enrichment (LEU). A typical research and test reactor was used, the MTR being simulated benchmark problems [9]. This reactor has an active core (height) of 60 cm, each combustible element has lateral dimensions of 7.7 cm X 8.1 cm. It is a pool type reactor, its fuel is composed of uranium oxide dispersed in Aluminum (UAl_x - Al) and power of 10 MW. The fuel element (EC) consists of 23 plates and the control / safety element (ECS) 21 plates being 17 identical to those of the EC and the others consisting of pure Al. Figure 1 provides the entire configuration of the reactor core.

As stated earlier, one of the input data sets of the WIMSD-5B code is relative to the core geometry and its material composition, information that is concatenated in the so-called equivalent cell.

4.1. Determination of the MTR-IAEA Equivalent Cell

The core of the MTR-IAEA reactor is heterogeneous, and one can identify a repeating geometric shape forming the active part of the nucleus. Considering a Fuel Element (EC) of the MTR-IAEA reactor, with its geometry and material composition shown in Figure 2 and Table 1, it is possible to associate with the EC an equivalent cell which is composed of four regions, as shown in Figure 4: region 1, representing the fuel or, also called crumb; the region 2, representing the coating of the fuel; region 3 representing the refrigerant and moderator and region 4 called the extra region, contaiing the inert edges of Al and excess water.

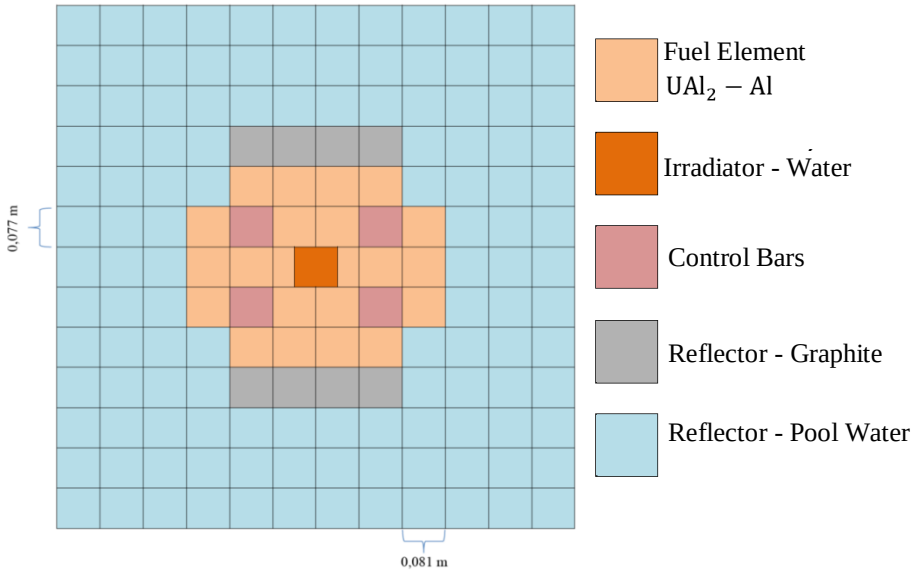


Figure 1: MTR-IAEA

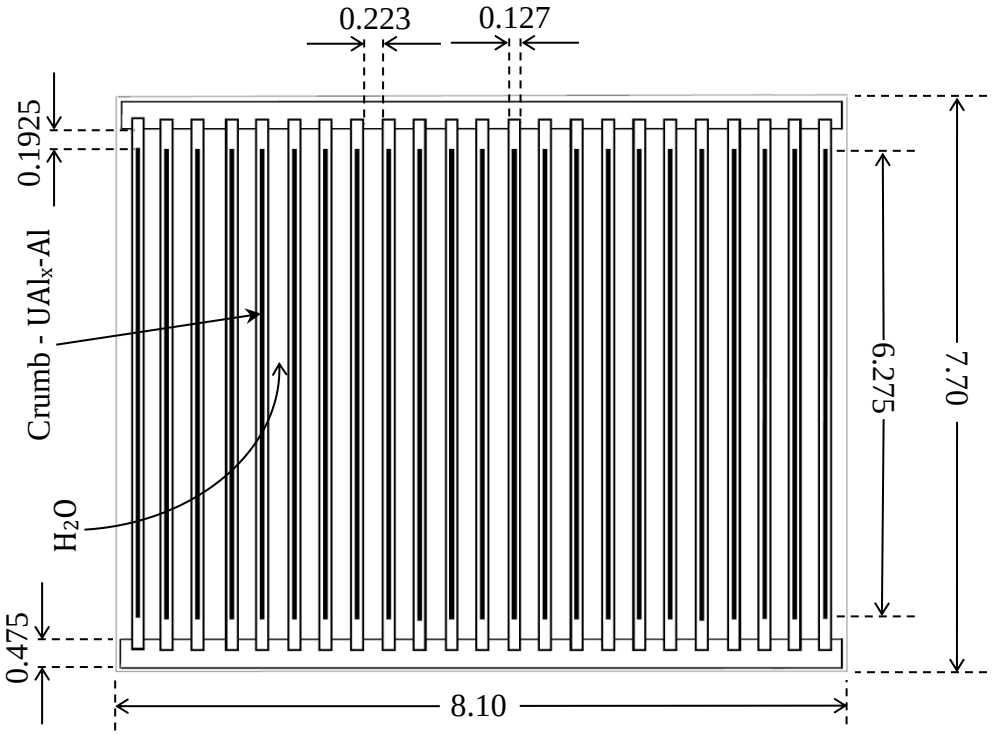


Figure 2: Cross section of the fuel element of the MTR-IAEA research reactor (units in cm).

Table 1: Dimensions and characteristics of the fuel element of the research reactor - MTR-IAEA

Fuel Type - UAl_x – Al
72w/o (72 % by weight) of Uranium in the UAl_x – Al
^{235}U powder content – 20 w/o (20 % by weight)
Mass of ^{235}U per EC (23 plates) - 390g
235U mass per plate – 16,95652g
It is considered only the ^{235}U and ^{238}U in fresh fuel
Dimensions of the Crumb – 6,275 cm x 0,051 cm x 60 cm

In order to obtain the equivalent cell initially the macro-cell M is formed from the straight section of an EC, its area being composed of the sum of the areas A and B, according to Figure 3. In the limits of this macrocell the net currents of neutrons are considered null. This

macrocell is representative for the entire reactor core. And then simple calculations of proportion between the areas and the atomic concentrations are performed.

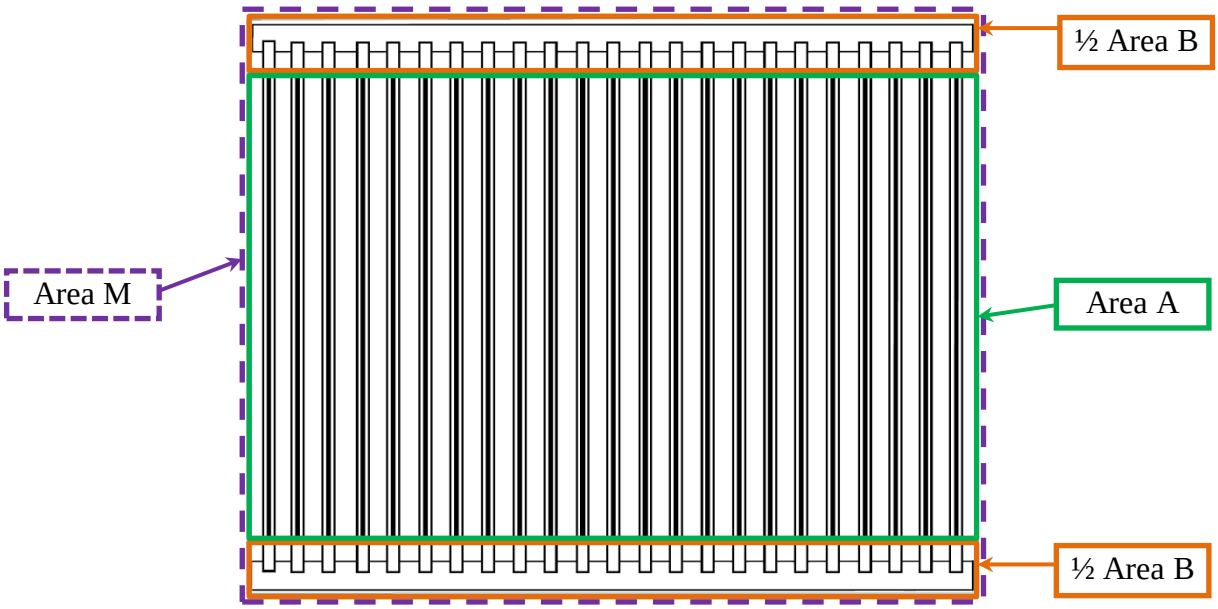


Figure 3: Macro cell M of the MTR-IAEA EC.

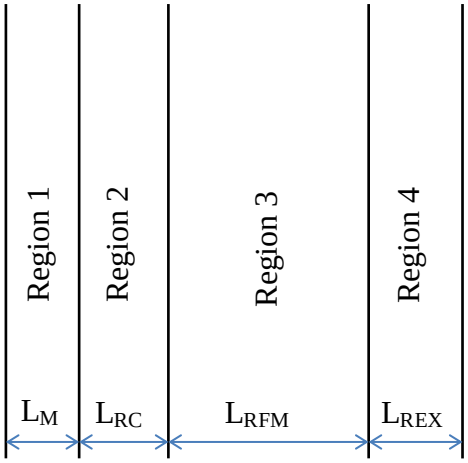


Figure 4: Mee definitive equivalent cell (non-scaled thickness).

4.2. Nuclear Data Libraries Evaluated Used

The other part of the "input" of the WIMSD-5B code is formed by the nuclear data library evaluated. For the simulations with the WIMSD-5B code, the following nuclear data libraries were used: IAEA, ENDFB-VII.1, JENDL3.2, JEFF3.1 and JEF2.2, for 69 energy groups and formatted by WLUP, WIMS Library Update Project.

For the IAEA, JENDL3.2, JEFF3.1 and JEF2.2 libraries, the WIMD-5B code in the burnup simulations provides the concentrations of 78 nuclides whereas for the library ENDFB-VII.1, it provides the concentrations of 89 nuclides. For comparison purposes, the concentrations of

the 78 nuclides corresponding to all libraries were considered, the 11 additional concentrations that the ENDFB-VII.1 library possessed were not considered.

Table 2: Atomic densities and thicknesses of the fuel element equivalent cell of the MTR-IAEA compared to the reference data.

Regions	Elements and Isotopes	Atomic Densities (atoms / barn.cm)		Thickness (cm)	
		Reference*	Calculated	Reference*	Calculated
Region 1: Crumb (UAl_xAl)	^{235}U	2.2536×10^{-3}	2.26271×10^{-3}	0.0255	0.0255
	^{238}U	8.9005×10^{-3}	8.93652×10^{-3}		
	Al	3.8171×10^{-2}	3.83265×10^{-2}		
Region 2: Coating	Al	6.0260×10^{-2}	6.02439×10^{-2}	0.038	0.038
Region 3: ½ Water Channel	H	6.6956×10^{-2}	6.66739×0^{-2}	0.1115	0.1115
	O	3.3428×10^{-2}	3.33695×10^{-2}		
Region 4: Inert Edges of Al and Excess Water	Al	no values available for comparison	4.60492×10^{-2}	0.0402	0.0397
	H		1.57441×10^{-2}		
	O		7.87207×10^{-3}		

(*) Source: [9].

4.3. Burn Simulation and Analysis

The burnup was simulated over a period of 400 days, which corresponds to a 50% reduction in the initial ^{235}U concentration. The variations during the burnig of the concentrations of some nuclides for the five nuclear data libraries evaluated were plotted in graphs and can be consulted in Figure 5. It can be seen that at the end of the simulation of the burning, in 400 days, the concentrations obtained using the five nuclear data libraries evaluated do not differ in most cases. However, not all libraries produced similar results in the following related cases: Cm-242, Cm-243, Dy-160, Dy-161, Dy-162, Dy-163, Figure 37, Ho-165, Er-166, Er-167, Ah-109, Te-127, I-127, Eu-152 and Am-242.

It is also important to point out the large variations in the concentrations in the first burnup steps (period few hours) in all the libraries and verified in the following cases: Np-239, Gd-157, Rh-105, Cd-113, I-135, Xe-135, MW-149, Sm-149, Sm-151 and U-237 PPF.

In addition to the variations of the nuclide concentrations, the results of the burnup simulations with the code WIMSD-5B, the variation of the infinite multiplication factor, k_∞, and the following nuclear parameters condensed for two energy groups were also obtained:

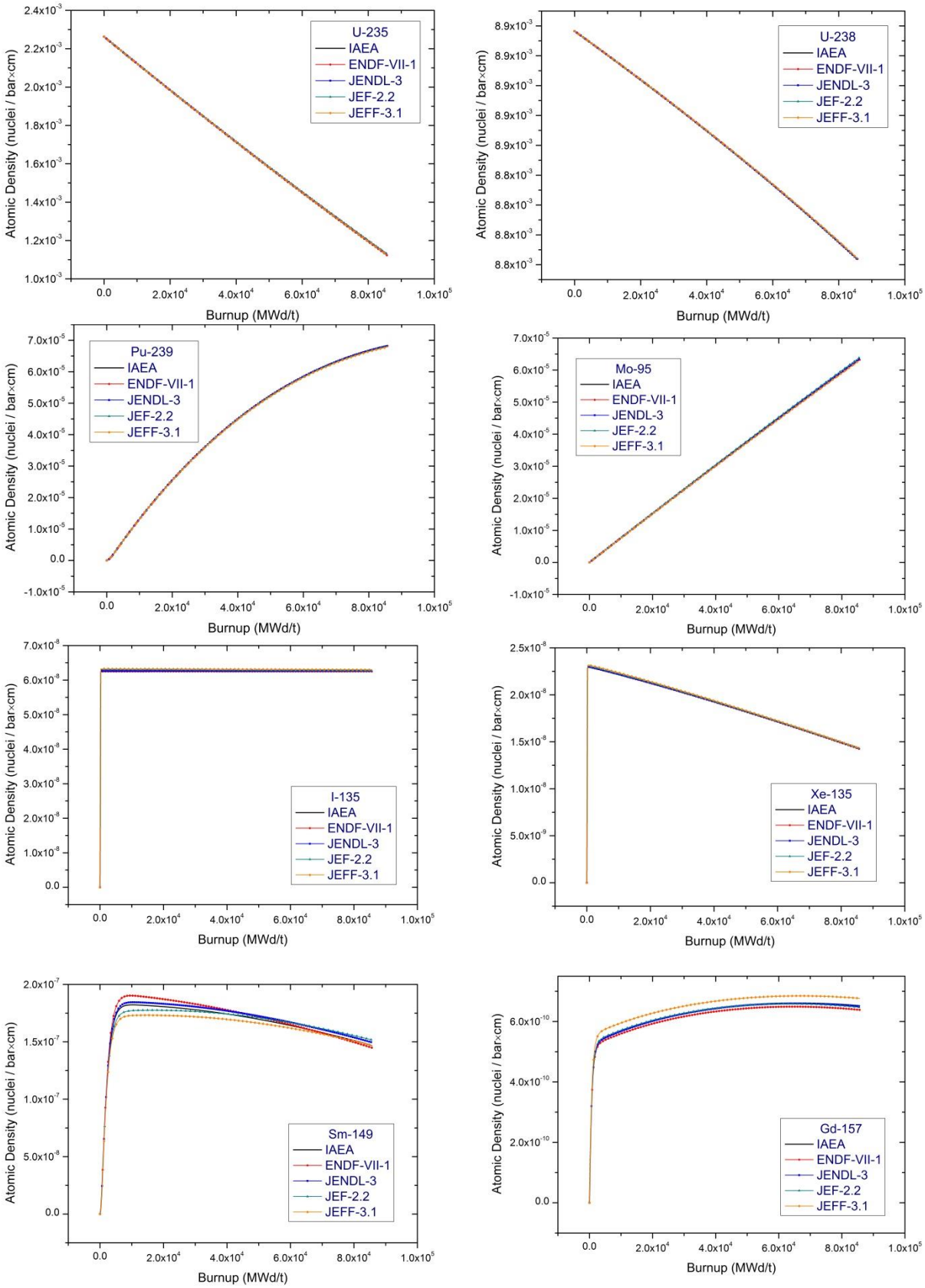


Figure 5: Variations during the burnup of the concentrations of some nuclides for the five nuclear data libraries evaluated.

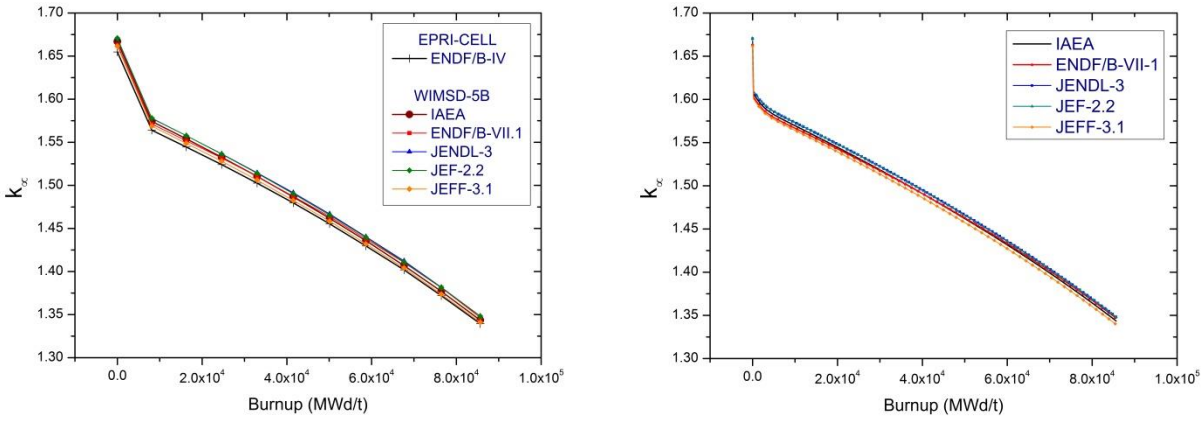
diffusion coefficients , D_g; absorption cross section, Σ_{ag}; removal cross section, Σ_{rg}; and the mean number of fission neutrons multiplied by the fission shock section, $\nu\Sigma_{fg}$.

In Figure 6a the plot of the k_∞ variation as a function of the burn for each nuclear data library is shown and compared to the result obtained with the EPRI-CELL code [9], which was used with the library ENDFB- IV. In this figure the graph was plotted with only a few points. In Figure 6b the graphic of the k_∞ variation as a function of burnup for each nuclear data library is shown in all burnup steps (400 days). Figure 7a show the graph plots of the nuclear group variations of the fast group, while Figure 7b show the plots of the graphs of the variations of the nuclear parameters of the thermal group.

It is seen that the variation of k_∞, Figure 6, is similar for each adopted library, and the same can be affirmed for $\nu\Sigma_{fg}$ in the two energy groups. However, the parameters D_g, Σ_{ag} and Σ_{rg} showed an offset in the curves according to the library used.

It is also observed that in the graph of Figure 6 the k_∞ varies accentuated form in the first burnup steps. The same can be observed in the thermal parameters of the thermal group.

These strong variations are directly related to the large increases in the concentrations of Xe-135 and Sm-149, Figure 5. Xe-135 is the main fission product that affects reactor operation due to its high absorption shock, in the amount of 2.65×10^6 barns. This isotope can be produced directly from the fission and also from the decay of other fission products. Sm-149 is another fission product of high absorption shock section, 41,000 barn.



(a) (b)

Figure 6: Variation of the infinite multiplication factor. (a) Comparison with code EPRI-CELL; (b) For all burnup steps.

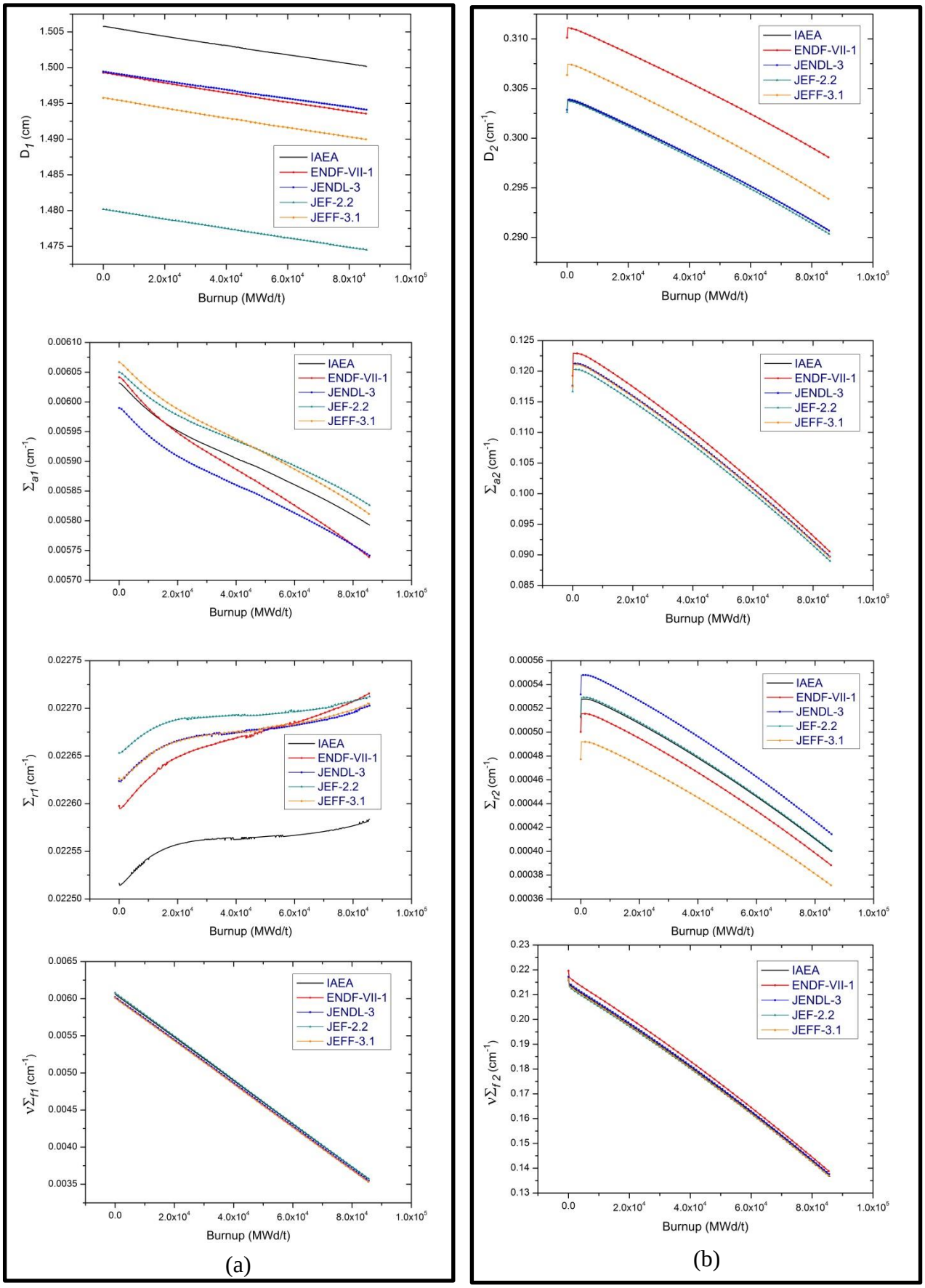


Figure 7: Variation of nuclear parameters. (a) Fast Group; (b) Thermal Group.

5. CONCLUSIONS

It was observed that in most cases the concentrations of nuclides presented practically the same result independently of the nuclear data library used.

The variation of the infinite multiplication factor was also observed and the variation of the parameter $\nu\Sigma_{fg}$ in the two energy groups was practically the same regardless of the nuclear data library used.

It is recommended for future work to use the evaluated nuclear data libraries for 192 power groups formatted by WLUP: IAEAGX, ENDFB-VII.1GX, JENDL3.2GX, JEFF3.1GX and JEF2.2GX.

It would also be appropriate to simulate the global calculations of the nucleus by solving the equation of neutron transport or diffusion using the homogenized nuclear parameters that were obtained for each of the nuclear data libraries evaluated and to verify how the neutron flux behaves in relation to each of the libraries.

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REFERENCES

1. KULIKOWSKA, T. Reactors Lattice Codes, Workshop on Nuclear Data and Nuclear Reactos: Physics, Design and Safety, Trieste, 13 março a 14 de abril 2000. Disponível em: http://users.ictp.it/~pub_off/lectures/lns005/Number_1/Kulikowska_3.pdf. 20/05/17.
2. INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA, WIMS-D LIBRARY UPDATE. Viena, 2007.
3. SILVA, C.A.M. Desenvolvimento de uma Metodologia para Simulação de Reatores Refrigerados a Gás com Propósito de Transmutação. UFMG, Belo Horizonte, 2009.
4. LESLIE, D. C. The 'SPECTROX' Method for Thermal Spectra in Lattice Cells. Reactor Science and Technology (Journal of Nuclear Energy Parts A/B, v. 17, p. 293-306, 1963.
5. CACUCI, D. G., Handbook of Nuclear Engineering, Volume 1, Springer, 2010.
6. HERMAN, M., TRKOV, A. (2009) ENDF-6 formats manual: data formats and procedures for the evaluated nuclear data file ENDF/B-VI and ENDF/B-VII. Technical report BNL-90365-2009, Brookhaven National Laboratory.
7. JEFF3 (2005) The JEFF-3.0 nuclear data library. Technical report JEFF Report 19, OECD Nuclear Energy Agency.
8. SHIBATA, K., HASEGAWA, A., IWAMOTO, O., et al (2002a) JENDL-3.2 covariance file. J. Nucl Sci. Technol (2):40–43.
9. INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA, Research Reactor Core Convertion from the Use of Highly Enriched Uranium to the use of Low Enriched Uranium Fuel, TECDOC-233-IAEA-(1980), Vienna.